Dose Equivalent Rate Evaluation for Nuclear Reactors Shielding Studies by Means of the Point Kernel Technique

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***Abstract-*A *shield* is a physical entity interposed between a source of ionizing radiation and an object to be protected such that the radiation level at the position of the object will be reduced. The term *shielding* refers to a system of shields assembled together. In nuclear reactor applications the radiations involved are fast and thermal neutron and gamma radiation (primary or secondary whether originated directly from the core or from neutron capture) . The dose equivalent rate to the target is given by the summation of the dose equivalent rate of each radiation component, which can be determined from the related radiation flux. As the Transport Equation requires substantial computational resources, simplified methods-such as the point kernel technique- have been developed for radiation flux calculation across shielding. In this work, the reliability of the point kernel technique for nuclear reactors radiation fields has been tested. Case studies regard three main types of nuclear reactors: Bulk Shielding Reactor, Light Water Reactor and Fast Breeder Reactor. In all cases, the point kernel overestimates radiation fluxes, and thus the dose equivalent rate, which turns to radiation protection safety purposes. Moreover, some observations have been drawn about the effectiveness of the proposed shielding in reducing the dose equivalent rate to the target to predefined limits, also according to the prevalent radiation type.**

Keywords-radiation shielding, point kernel, dose equivalent rate, dose calculation, reactor shielding.

I. INTRODUCTION

A shield is a physical entity interposed between a source of ionizing radiation and an object to be protected such that the radiation level at the position of the object will be reduced. The term shielding refers to a system of shields assembled for a specific radiation protection purpose. Shielding design requires a thorough knowledge on radiation sources and materials characteristics as well as radiation-matter interaction phenomena.

Particle flux can be obtained by solving the Transport Equation, whose complexity requires substantial computational resources. Therefore, approximate methods, such as the point kernel technique, are usually applied. The uncertainties entailed by the rough simplifications on which these methods are based are irrelevant to radiation protection purposes, since it is often enough to determine only the order of magnitude of the dosimetric quantity.

The aim of this work is to apply the point kernel technique to some practical cases concerning nuclear reactors and to evaluate its limitations in order to establish the reliability of such method as an alternative to more complex computational codes for particle transport.

Case studies have been analysed according to two different approaches:

- Verification issue: determination of dose downstream of an already existing shielding;
- Design issue: determination of the thickness of a shielding's layer in order to limit the dose to a predetermined value.

If available, results have been compared to real data.

II. RADIATION SOURCES

Radiation fields are extremely difficult to study due to changes of its initial features by its interaction with matter. As a matter of fact, a neutron radiation field gives rise to a gamma field by neutron capture (n,γ); more specifically, in case of thermal neutron capture the gamma radiation that derives may be quite penetrating to the extent of becoming the major contributor to the radiation field. Hence, main importance is given to material selection and set-up.

The radiation fields taken into account are:

- Fast neutron radiation: coming directly from the source;
- Thermal neutron radiation: originated in the shielding by fast neutrons thermalization;
- Primary gamma radiation: coming directly from the source;
- Secondary gamma radiation: due to thermal neutron capture in the shielding.

A. *Point Kernel Technique*

The point kernel K is a mathematical relation that provides the hypothetical response of a detector located at a distance R from a single isotropic point source in an infinite and homogeneous medium. One of its main features is the possibility of evaluating the effects of radiation beamed from an extended source. In fact, the point kernel definition is based on the assumption that any radiation source can be seen as consisting of differential isotropic point sources. The overall effect of radiation is given by the integration (or summation) of the contributions from each individual differential source into which the region of interest has been divided.

The point kernel method has been firstly developed for photon radiation shielding. Despite the underlying physical

principal is different, the same analytical formulation is used for fast neutron radiation.

The expression used, considering also the uncollided particle flux, is:

$$K(R) = \frac{1}{4\pi R^2} B(R,\mu) e^{-\mu R} \quad (1)$$

where μ is the attenuation coefficient and B is the buildup factor.

The strategy behind photon shielding is to place a material with high atomic number (Z) between the source and the target, so as to promote the photoelectric effect and thus, the complete removal of the photon from the beam.

B. *Removal Cross Section*

On the other hand, neutron shielding is accomplished by:

- Removing fast neutrons from the uncollided beam through elastic and inelastic scattering;
- Thermalization of the "removed neutrons" at point (1);
- Neutron diffusion and absorption.
- Scattering interactions during slowdown may divert the neutron's direction, according to the scattering angleθ, such that some neutrons may not reach the detector. As a matter of fact, the scattering effect is that of deducting a fraction of neutrons from the beam due to the deviation from collision against the shield's heavy nuclei.

According to the Removal Theory, the fraction of removed neutrons from the uncollided beam can be expressed through the *removal cross section (RCS), as:*

$$\Sigma_R(E) = \Sigma_{tot}(E) - \overline{\psi}(E)\Sigma_{es}(E) \quad (2)$$

Where $\Sigma_R(E)$, $\Sigma_{tot}(E)$ and $\Sigma_{es}(E)$ are respectively the macroscopic removal, total and elastic scattering cross sections, and $\overline{\psi}(E)$ the scattering mean cosine in the laboratory system. The following simplified expression for neutron attenuation is used:

$$\varphi(R) = \frac{1}{4\pi R^2} e^{-\sum_k \Sigma_R^{(k)}(E) z_k} \quad (3)$$

which indicates the mean fast neutron flux that passes through a multilayer shielding at a distance R from the single point source. $\Sigma_R^{(k)}$ and z_k are respectively the RCS and the thickness of the kth-layer.

III. RADIATION FLUX EVALUATION

Basic calculations are performed assuming an infinite slab model, and then corrections are applied to account for the quasi-spherical or quasi-cylindrical shape of the actual system. Geometric correction factors are the following:

$$\begin{aligned} g_c &= \left(\frac{r_0}{r}\right)^{\frac{1}{2}} && \text{for cylindrical geometries} \\ g_c &= \frac{r_0}{r} && \text{for spherical geometries} \end{aligned} \quad (4)$$

where r_0 is the radius of the sphere or of the cylinder surface and r is the measuring distance.

A. *Fast Neutron Flux Evaluation*

The volume source strength of fast neutrons in a reactor core is:

$$S_V^n = 7.8 \cdot 10^{16} \frac{P}{V} \quad [\text{cm}^{-3}\text{s}^{-1}] \quad (5)$$

where P is the maximum operating power of the reactor [MW] and V is its volume. The fast neutron flux at the target is given by:

$$\varphi_f = \frac{S_V^n}{2\overline{\Sigma}_R}[E_2(b_1) - E_2(b_2)] \quad (6)$$

where

- $\overline{\Sigma}_R$ is the average removal cross section for the core materials;
- $b_1 = \sum_i \Sigma_R^{(i)} t_i$ is the sum of the mean free paths throughout all core materials;
- $\Sigma_R^{(i)}$ and t_i are, respectively, the neutron removal cross section and the width of the kth-layer;
- $b_2 = 2\overline{\Sigma}_R R_c$;
- R_C is the core radius;
- $E_2(b)$ is the exponential integral function of order 2.

C. *Thermal Neutron Flux Evaluation*

The thermal neutron flux is obtained from the fast neutron flux, as follows:

$$\varphi_{th}(x) = B_{th}\varphi_f(x) \quad (7)$$

and

$$B_{th} = \frac{\Sigma_R \exp(\Sigma_R^2 \tau - h)}{\Sigma_a[1 - (\Sigma_R L)^2]} \quad for\ L < \frac{1}{\Sigma_R} \quad (8)$$

B. *Primary Gamma Ray Flux Evaluation*

The primary gamma volume source strength for photons of energy E is:

$$S_V^\gamma = 3.11 \cdot 10^{16} \eta_p(E) \frac{P}{V} \quad [\text{cm}^{-3}\text{s}^{-1}] \quad (9)$$

where $\eta_p(E)$ is the number of photon emitted per fission into the energy band designated by E. The total gamma ray flux at P is given by:

$$\varphi_\gamma^p = \frac{S_V^\gamma}{2\mu_s} \bar{\alpha} e^{-b_1}[1 - e^{-2\mu_s R_C}] \quad (10)$$

In which the linear buildup factor has been already considered as $B = \alpha\mu t$, and being:

- $b_1 = \sum_k \mu_k t_k$ is the sum of the mean free paths of all layers constituting the shielding, or attenuating thickness of the shield;

- μ_k e t_k are, respectively the linear attenuation coefficient and the thickness of the k^th-layer;
- μ_s is the linear attenuation coefficient of the core region averaged over core materials;
- $\bar{\alpha}$ is the average build-up factor parameter for the shielding (a linear expression for the buildup factor has been considered only as an hypotheses as the real flux trend is unknown).

If the buildup factor is expressed by Taylor's formula, eq. (4.3_2) turns into the following relation:

$$\varphi_\gamma^p = \frac{S_V^\gamma}{2\mu_s}\sum_{i=1}^{n} A_i\{E_2[(1+\alpha_i)b_1] - E_2[(1+\alpha_i)b_1 + b_2]\} \quad (11)$$

$$B(t,\mu) = \sum_{i=1}^{n} A_i e^{-\alpha_i \mu t} \quad where \sum_i A_i = 1 \quad (12)$$

C. Secondary Gamma Ray Flux Evaluation

The volume source strength of secondary gamma rays depends on the point (x) at which the thermal neutron flux has been previously calculated, as secondary gamma rays are generated principally by the thermal neutrons capture:

$$S_V^{\gamma S}(x) = \eta_s(E)\Sigma_C^{(j)} S_k e^{-\lambda_k x} \quad (13)$$

where:

- $\eta_s(E)$ is the number of photons emitted with energy E per neutron capture in the generic point x within the k^th-layer;
- $\Sigma_C^{(k)}$ is the macroscopic thermal neutron capture cross section for the material of the k^th-layer;.
- $S_k e^{-\lambda_k x}$ indicates the thermal neutron flux φ_{th} in the k^th-layer.

The total secondary gamma ray flux at the target, due to the k^th-layer which contains an exponentially varying source $S_V^{\gamma S}$ is given by eq. (4.4_2).

$$\varphi_\gamma^s = \frac{\eta_s(E)\Sigma_C S_k}{2}\,\bar{\alpha}\left[\frac{e^{-\mu_k t_k} - e^{-\lambda_k t_k}}{\lambda_k - \mu_k}\right] e^{b_1} \quad (14)$$

where $b_1 = \sum_{i=k+1}^{n} \mu_i t_i$.

D. Dose Calculation

The dose equivalent rate can be obtained for each radiation contribution by applying an appropriate dose rate conversion factor f_c-specific for each kind of radiation- to the related flux,

$$[f_C] = \frac{\text{mrem/h}}{\text{cm}^{-2}\text{s}^{-1}}$$

The values for f_c can be found in charts and graphs [6]. The overall aggregate dose equivalent rate is given by the summation of the contributions from the fast neutron and thermal neutron fluxes, primary gamma rays and secondary gamma rays:

$$\dot{H} = \dot{H}_f + \dot{H}_{th} + \dot{H}_\gamma^p + \dot{H}_\gamma^s$$

By means of the conversion factors in ICRP publication n. 74 it is possible to obtain the effective dose rate as well.

IV. CASE STUDIES

The mathematical expressions quoted above are applied to some significant case studies, concerning different kinds of nuclear reactors. Its interest relies in the entireness of radiation contributions.

All the following calculations are referred to an AP irradiation.

A. Bulk Shielding Reactor

The Bulk Shielding Reactor (BSR) is a pool-type reactor with a parallelepiped core shielded in water. The core's shape can be approximated to a cylinder of radius 21 cm and of volume $1.064 \cdot 10^5$ cm^3. The aim of this calculation is to compare the results obtained with the point kernel technique for the fast neutron flux φ_f and for the thermal neutron flux φ_{th} with the experimental values for different water thicknesses. Furthermore, the dose equivalent rate $\dot{H}$ due to all four radiation contributions is determined.

All results are normalized to a reactor power of 1 watt. Table I shows the main parameters needed in calculation.

TABLE I- BSR INPUT DATA FOR CALCULATION.

Fermi age [cm^2]	31.4
Diffusion length [cm]	2.27
Macroscopic capture cross section for water [cm^{-1}]	0.0196
Macroscopic removal cross section for water [cm^{-1}]	0.10
Macroscopic removal cross section averaged over core materials [cm^{-1}]	0.096
Correction factor h [-]	1.3
Conversion factor for fast neutrons f_c [(mrem/h)/(cm^{-2} s^{-1})]	0.147
Conversion factor for thermal neutrons f_c [(mrem/h)/(cm^{-2} s^{-1})]	0.00384

The expressions used are: (4.1_2), (4.2_1), (4.3_2), (4.4_2).

B. Light Water Reactor (LWR)

The purpose is to ascertain that a 24 cm thick layer of lead is enough to reduce the dose rate at the outer surface of the radial shielding to 6mrem/h. The reactor's main characteristics are listed in Table II.

TABLE II- LWR MAIN CHARACTERISTICS

Thermal Power	**10 MW**		
Core	Dimensions	Radius [cm]	50
		Height [cm]	100
	Composition	H_2O	29.44%
		Zirconium	70.55%
		Uranium	0.01%
Vessel	Layer no.	Material	Thickness [cm]
	1	H_2O	35
	2	Steel	25
	3	H_2O	115
Shield	4	Pb	24

Data needed for the neutron buildup factor evaluation is shown in Table III. The continuous spectrum of the primary gamma rays can be discretized into five photon groups, reported in Table IV.

TABLE III- LWR BUILDUP FACTOR PARAMETERS

Material	$\boldsymbol{\Sigma_R}$	$\boldsymbol{\Sigma_a}$	**L**	$\boldsymbol{\tau}$	**h**
H_2O	0.098	0.0111	2.76	31.4	0.2
Fe	0.166	0.1268	1.33	200	5.5
Pb	0.118	0.0056	13.6	5000	73

TABLE IV – LWR: NUMBER OF PRIMARY GAMMA RAYS EMITTED FOR DIFFERENT ENERGY GROUPS.

Photon energy $\boldsymbol{\Delta E}$ [MeV]	$\boldsymbol{\eta_P}$
0-1	12.0
1-3	5.31
3-5	1.02
5-7	0.527
7-9	0.125

TABLE V- FBR MAIN CHARACTERISTICS

Reactor power	**500 MW**		
Core dimensions	Radius [cm]	42	
	Height [cm]	78	
Core composition	^{239}Pu	10 %	
	^{238}U	20 %	
	^{56}Fe	25 %	
	^{23}Na	45 %	
Shielding	Layer no.	Material	Thickness [cm]
	1	^{238}U	60
	2	^{23}Na	35
	3	^{56}Fe	25
	4	Graphite	100
	5	Concrete	95.5

The energies of secondary gamma rays are 2.2 MeV, 7.0 MeV, 8.0 MeV, which are emitted respectively from water, iron and lead. For all three materials the emission of one photon per thermal neutron captured is assumed (η_s=1).

The attenuation factors, as well as the coefficients *A* and α in Taylor's buildup formula, can be readily deduced from basic data given in shielding manuals.

The expressions used for radiation fluxes are: (4.1_2), (4.2_1), (4.3_3) e (4.4_2).

C. Fast Breeder Reactor

The reactor examined is a prototype sodium cooled fast reactor, having the basic properties summarized in Table V. The dose equivalent rate at the outer surface of the shielding is calculated. Other data to be used in calculation is shown in Table VI.

A distinctive feature of a fast breeder reactor is the presence of a blanket of fertile material surrounding the core. To simplify calculation fast fission events in the blanket are neglected, so that the blanket region is considered as part of the shielding. Also, in the sodium region next to the ^{238}U blanket gamma rays from activated sodium atoms (^{24}Na) are disregarded: the absorption rate of neutrons in the sodium (core + blanket) which produces ^{24}Na atoms is negligible.

The primary gamma rays are:

- Prompt fission gamma rays;
- Fission products gamma rays;
- ^{238}U and ^{239}Pu capture gamma rays;
- Stainless steel capture on the vessel;
- ^{23}Na capture.

TABLE VI- BUILDUP FACTOR PARAMETERS FOR EACH LAYER.

Material	$\boldsymbol{\Sigma_R}$	**L**	$\boldsymbol{\tau}$	**h**	$\boldsymbol{\Sigma_a}$
^{238}U	0.174	1.51	4000	120	0.13
^{23}Na	0.032	16	7000	6	0.013
^{56}Fe	0.166	1.33	200	2.8	0.21
Graphite	0.065	52	364	0	2.60E-04
Concrete	0.08	8.9	179	1.5	9.00E-03

TABLE VII- NUMBER OF PRIMARY AND SECONDARY GAMMA RAYS EMITTED.

Photon energy [MeV]	0.5	2	4	6	
η_p	11.62	5.04	0.61	0.095	
Photon energy [MeV]	2	3	7	5	8
Material	U	Na	Fe	C	concrete
η_s	2	4	1	1	1

On the other hand, in secondary gamma flux calculation each layer is considered singularly as a radiation source. It is necessary to evaluate the thermal neutron flux in a preliminary step, from whose attenuation the number of captured neutrons at each layer- and, ultimately, the number of gamma rays emitted- can be estimated.

The procedure is the following:

- Evaluation of the fast neutron flux at the interface between layers *i* and *i+1*;
- The thermal neutron flux at each layer is obtained by applying to the fast neutron flux above the buildup factor corresponding to the layer's material;
- Calculation of the exponential attenuation factor λ_i from: $\varphi_{th}^{i} = \varphi_{th}^{i-1}\exp(-\lambda_i z)$, where z is thickness of the i layer;
- Secondary gamma flux is determined from (4.4_2).

The expressions used for the remaining radiation fluxes are: (4.1_2), (4.2_1), (4.3_2).

V. RESULTS

A. Bulk Shielding Reactor

Equations (4.1_2) e (4.2_1) have been rectified with the correction factor g_c. The values for the fast and thermal neutron fluxes with distance in water s_w are given in Table VIII. Fig. 1 and Fig. 2 show the trend of fluxes calculated with the point kernel technique $\varphi^{(c)}$ in comparison with experimental results$\varphi^{(m)}$. The neutron buildup factor equals to 2.06.

As regards gamma secondary flux, the *S* e λ factors are respectively $5.20{\cdot}10^5$ cm^{-2}s^{-1} and 0.115. Table X summarizes the primary gamma flux for different energy values and the

secondary gamma flux. Table XI shows the dose equivalent rate (d.e.r.) related for all components with distance.

Both the neutron fluxes (thermal and fast) obtained by point kernel calculations are always lower than the experimentally measured values. As regards thermal neutron flux, the bias increases with distance from the shielding. Major contribution to the dose equivalent rate is due to gamma radiation.

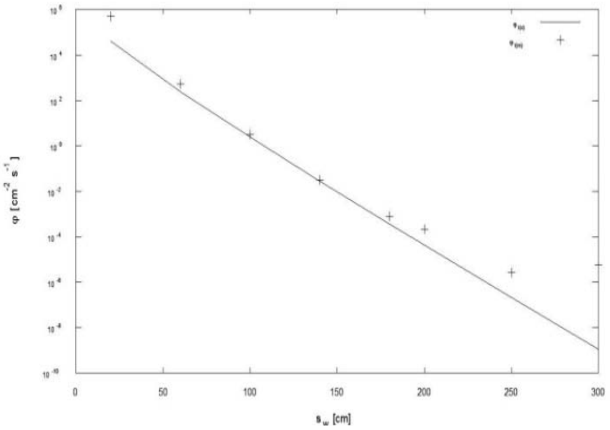

Fig. 1 – experimentally measured thermal neutron flux $\varphi_{th}^{(m)}$ vs. point kernel thermal neutron flux $\varphi_{th}^{(c)}$.

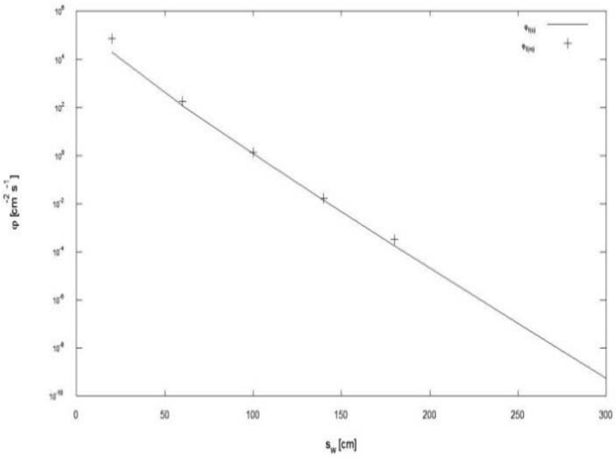

Fig. 2 - experimentally measured fast neutron flux $\varphi_{f}^{(m)}$ vs. point kernel fast neutron flux $\varphi_{f}^{(c)}$.

TABLE VII – BSR: EVALUATED THERMAL AND FAST NEUTRON FLUX WITH DISTANCE.

s_w [cm]	$\varphi_{th}^{(c)}$	$\varphi_{f}^{(c)}$
20	4.03E+04	1.95E+04
60	2.42E+02	117.4
100	2.38E+00	1.154
140	2.81E-02	0.01366
180	3.68E-04	1.79E-04
200	4.31E-05	2.09E-05
250	2.13E-07	1.04E-07
300	1.11E-09	5.41E-10

TABLE IX – BSR: NUMBER OF PRIMARY GAMMA RAYS EMITTED FOR DIFFERENT ENERGY GROUPS.

Photon Energy [MeV]	1.0	2.0	4.0	6.0	8.0
η_P	8.61	2.705	0.413	0.065	0.022

TABLE X- BSR PRIMARY AND SECONDARY GAMMA FLUXES PER ENERGY GROUP AND WATER LAYER THICKNESS.

s_w	$\varphi_\gamma^{(P)}$ [MeV]					$\varphi_\gamma^{(S)}$
[cm]	1	2	4	6	8	2.2
20	9.56 E+6	2.40 E+6	3.35 E+5	5.79 E+4	1.49 E+4	1.95 E+4
60	5.68 E+5	3.33 E+5	8.52 E+4	1.60 E+4	5685	2722
100	3.37 E+4	4.64 E+4	2.20 E+4	5308	2177	334.6
140	2001	6458	5656	1767	833.5	42.92
180	118.8	898.7	1457	588.1	319	5.68
200	28.95	335.3	739.8	339.3	197.5	2.08
250	0.848	28.5	135.8	85.79	59.48	0.172
300	0.0248	2.423	24.94	21.69	17.91	0.0145

TABLE XI1- BSR D.E.R. FOR EACH RADIATION COMPONENT AND WATER LAYER THICKNESS.

s_w	$\dot{H}_f$	$\dot{H}_{th}$	$\dot{H}_\gamma^{(p)}$	$\dot{H}_\gamma^{(S)}$
20	2871	154.9	2.64E+4	62.33
60	17.1	0.931	2.58E+3	8.71
100	0.17	9.16E-3	3.55E+2	1.231
140	2.01E-3	1.08E-4	6.78E+1	1.37E-1
180	2.63E-5	1.42E-6	1.59E+1	1.82E-2
200	3.08E-6	1.66E-7	8.29	6.66E-3
250	1.52E-8	8.22E-10	1.74	5.50E-4
300	7.95E-11	4.29E-12	4.04E-1	4.64E-5

B. Light Water Reactor (LWR)

The geometric correction factor at a distance of 24 cm from the core (source) is 0.448. The macroscopic core removal cross section is determined as the weighted average over the cross sections of core materials, and equals to 0.1 cm^{-1}. The shielding buildup factor has been determined in the same manner.

TABLE XII-LWR NEUTRON FLUXES AND RELATED D.E.R.

Fast Neutrons		**Thermal Neutrons** $\bar{B} = 8.24$	
φ_f [$cm^{-2}s^{-1}$]	$\dot{H}_f$ [mrem/h]	φ_{th} [$cm^{-2}s^{-1}$]	$\dot{H}_{th}$ [mrem/h]
36.22	5.07	298.5	1.15

Given its high value of Z, he shield buildup factor for gamma radiation matches the buildup factor for the lead layer. The primary gamma flux for different energy groups- and the related dose equivalent rate- are shown in Table XIII. The total dose equivalent rate due to primary gamma radiation $\dot{H}^{p}_{\gamma,tot}$ is 0.5631 mrem/h. Results for secondary gamma flux are listed in Table XIV, as well as data needed for its calculation.

TABLE XIII- LWR PRIMARY GAMMA FLUX PER ENERGY GROUP AND RELATED D.E.R.

E [MeV]	φ^{p}_{γ} **[cm^{-2}s^{-1}]**	$\dot{H}^{p}_{\gamma}$ **[mrem/h]**
0.5	*negligible*	*negligible*
2	11.51	0.0368
4	45.93	0.2297
6	9.87	0.0592
8	26.68	0.2374

TABLE XIV- LWR SECONDARY GAMMA FLUX FOR EACH LAYER AND RELATED D.E.R.

Layer	**Material**	S	λ	**E [MeV]**	φ^{s}_{γ} **[cm^{-2}s^{-1}]**	$\dot{H}^{s}_{\gamma}$ **[mrem/h]**
1	H_2O	5.97 E+12	0.0904	2.2	5.29 E-11	Negl.
2	Steel	2.52 E+11	0.276	7	135.4	0.9478
3	H_2O	2.53 E+8	0.094	2.2	0.163	4.597 E-4
4	Pb	7700	0.135	8	2.013	0.01610

The overall dose equivalent rate is 7.747 mrem/h, which is greater than the threshold value ordained. Therefore, the thickness of the lead layer is augmented in order to limit the equivalent dose rate to 6 mrem/h. Optimization is executed by limiting the share of only the major contributor to the dose equivalent rate, which is the fast neutron flux

(Fig. 3). Given the share of each radiation component, neutron flux contribution to a d.e.r. of 6mrem/h should be 3.926 mrem/h, which corresponds to a maximum acceptable neutron flux of 28.04 cm^{-2}s^{-1}.

Making the reverse calculations, the minimum thickness for the lead layer is 26.5 cm. The d.e.r. for all four radiation components in this new set-up are shown in Table XV.

The total dose equivalent rate for the new arrangement in 5.663 mrem/h or an effective dose rate of 0.0142 μSv/s [6]).

C. Fast Breader Reactor

The shielding's geometric correction factor is 0.343. The neutron buildup factor for the entire shield has been calculated as in the cases above. Table XVI shows results for fast and thermal neutron fluxes as well as the related d.e.r. The share of primary gamma radiation to d.e.r. is negligible, as of order of magnitude 10^{-18} (or even lower).

The total equivalent dose rate is 2.00 mrem/h, which corresponds to an effective dose rate of $5.067 \cdot 10^{-3}$ μSv/h [6]. The major contribution to the d.e.r. comes from the secondary gamma flux- accounting for 98%-, produced from iron in the steel layer (7 MeV). Although gamma rays originated in the concrete layer are of higher energy (8 MeV), they affect to a lesser degree as the thermal neutron flux at the last layer has been highly attenuated and as self-absorption in concrete is highly probable.

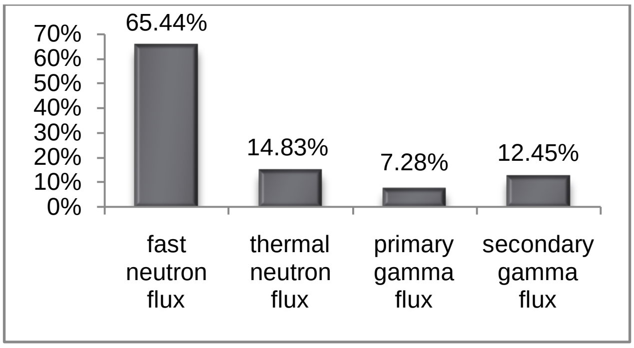

Fig.3- radiation componenents share to the total d.e.r.

TABLE XV2- LWR: D.E.R. FOR RADIATION COMPONENTS AFTER SHIELD OPTIMIZATION.

$\dot{H}_f$	$\dot{H}_{th}$	$\dot{H}^{p}_{\gamma,tot}$	$\dot{H}^{s}_{\gamma,tot}$
3.702	1.340	0.3503	0.2705

TABLE XVI-FBR NEUTRON FLUXES AND RELATED D.E.R.

Fast Neutrons		**Thermal Neutrons** $\bar{B} = 7.151$	
φ_f [cm^{-2}s^{-1}]	$\dot{H}_f$ [mrem/h]	φ_{th} [cm^{-2}s^{-1}]	$\dot{H}_{th}$ [mrem/h]
0.481	0.067	3.44	0.013

TABLE XVII – FBR SECONDARY GAMMA FLUX FOR EACH LAYER AND RELATED D.E.R.

Material	**E [MeV]**	η_s	$\varphi^{s}_{\gamma,i}$ **[cm^{-2}s^{-1}]**	$\dot{H}^{s}_{\gamma,i}$ **[mrem/h]**
^{238}U	2.00	2	*Negligible*	*Negligible*
^{23}Na	3.00	4	0.071	2.78E-04
^{56}Fe	7.00	1	272.775	1.92
Graphite	5.00	1	0.346	1.92E-03
Concrete	8.00	1	3.906	3.03E-02

AKNOWLEDGMENTS

The point kernel technique has revealed itself as a useful tool to apply in those cases in which computer codes cannot be run. Main issues concern calculation of mixed radiation fields. In particular, the evaluation of the thermal neutron flux from the fast one through an appropriate buildup factor arises the complexity on accurately estimating the latter, as well as the inability to define the buildup factor immediately close to the source surface. Furthermore, thermal neutron flux calculated with this method differs increasingly from experimental values with distance from the source.

Lead has been noted to be a suitable material for gamma radiation shielding, even with little thicknesses, but not equally for thermal neutrons attenuation. In fact, thermal neutron capture in lead originates a (n,γ) reaction, with emission of an 8 MeV gamma ray. On the other hand, fast neutrons can be simply attenuated by interposing a layer of water- like in reactor applications between the barrel and the reactor's vessel- or other hydrogenate material. Downstream of this layer, it is advisable to place a layer of a material capable of capturing the neutrons just thermalized.

All considerations above suggest that the shielding proposed for the BSR is effective for fast neutron flux attenuation, but not for gamma radiation (especially, secondary gamma flux) to the same extent. On the contrary, the LWR shielding described is appropriate for gamma radiation, as it contains a lead layer, whereas the fast neutron component is not sufficiently attenuated due to a lack of scattering material. Finally, the FBR shielding is capable of capturing both fast and thermal neutrons, but gamma radiation reaches the outer surface as no high atomic number material has been provided.

In conclusion, it has been observed that the point kernel technique overestimates the radiation fluxes, which represents an excellent feature that turns to radiation protection's advantage. Point kernel results are much more accurate as further away the target is from the source. This characteristic derives directly from the definition of removal cross section.

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